NON-PUBLIC?: N
ACCESSION #: 9312210195
LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK-UNIT 2 PAGE: 1 OF 7

DOCKET NUMBER: 05000446

TITLE: LOOSE JAM NUT ON THE POSITIONER FEEDBACK LINKAGE ARM OF SEPARATOR DRAIN TANK NORMAL VALVE CAUSED AN ESF ACTUATION
EVENT DATE: 11/17/93 LER #: 93-011-00 REPORT DATE: 12/15/93

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: W. G. GULDEMOND, MANAGER, TELEPHONE: (817) 897-8739
SYSTEM ENGINEER

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On November 17, 1993, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1. Power operation with reactor power at 100 percent.

At 7:59 p.m. on November 17, 1993 Separator Drain Tank, 2A Normal Drain Valve failed open. The failure of the valve was caused due to the jam nut on the Fisher Positioner feedback linkage arm being loose. The jam nut was apparently loosened due to vibration.

The corrective actions were to tighten the jam nut, inspect other similar and identical feedback positioner arms and apply similar corrective actions as applicable.

END OF ABSTRACT

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)(EIIS:(JC)).

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On November 17, 1993, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in MODE 1, Power Operation, with reactor power at 100 percent.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems, or components that contributed to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

At 7:58 p.m. on November 17, 1993, the Separator Drain Tank 2A (EIIS:(TK)(SN)) Normal Drain Valve 2-LV-2708 (EIIS:(V)(SN)) failed open. With valve 2-LV-2708 open, pressure in Heater Drain Tank (HDT) 2-01 began to increase. The pressure in HDT 2-01 quickly equalized with the pressure in HDT 2-02, forcing water out of HDT 2-01 and into HDT 2-02. As a result, the level in HDT 2-01 decreased to approximately 40 percent, while the level in HDT 2-02 rose to 90 percent, opening the HDT Alternate Drain Valve 2-LV-2594 (EIIS:(V)(SN)). Valve 2-LV-2594 controls level in both HDTs via a common discharge line.

With valve 2-LV-2594 open, level in both HDT 2-01 and HDT 2-02 decreased. The level in HDT 2-01 decreased to approximately 28 percent, while HDT 2-02 was approximately 50 percent. At this point, valve 2-LV-2592 (EIIS:(V)(SN)), the Heater Drain Pump discharge valve, closed on low level in HDT 2-01. This valve directs flow from the discharge of the Heater Drain Pumps to the suction of the Main Feedwater Pumps (MFP) (EIIS:(P)(SJ)).

With the discharge valve closed, the Heater Drain Pumps went on full flow recirculation back to the HDTs.

At 7:59 p.m. on November 17, 1993, the Reactor Operator (RO) (utility, licensed) noticed MFP suction pressure decreasing and immediately began to reduce turbine load. MFP 2-B tripped on low suction pressure, resulting in an automatic turbine runback. The low suction pressure was caused by the loss of forward flow from the Heater Drain Pumps. The RO placed the control rods in AUTO, witnessed them inserting, then commenced boration. With the loss of MFP 2-B, flow to the Steam Generators (SG) (EIIS:(SG)(SB)) was reduced. The RO verified that the Feedwater Flow Control valves were open and began to monitor SG levels. SG levels continued to decrease until finally at 8:01 p.m. on November 17, 1993, a Unit 2 reactor trip occurred due to LO LO Level in SG No. 2. The RO immediately entered the Emergency Operating Procedures (EOP) and began recovery actions.

At 9:12 p.m. on November 17, 1993, with recovery actions still in progress, MFP 2-A tripped for no apparent reason. As a result, another ESF actuation occurred, causing the Feedwater Flow Control valves to trip open. Attempts to reset MFP 2-A failed. Plant recovery actions continued with MDAFW pumps in service. At 9:40 p.m. on November 17, 1993, CPSES Unit 2 was stabilized in MODE 3, Hot Standby.

An event or condition that results in an automatic or manual actuation of any ESF, including the RPS, is reportable within 4 hours under 10CFR50.72(b)(2)(ii). At 9:40 p.m. on November 17, 1993, the Nuclear Regulatory Commission Operations Center was notified of the event via the Emergency Notification System.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT FAILURE, OR PROCEDURAL OR PERSONNEL ERROR

An alarm on the Main Control Board alerted the RO for both the actuations. The RO took immediate corrective action but could not maintain SG levels. The reactor tripped at 8:01 p.m. on November 17, 1993, on LO LO Level in SG No. 2.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

Normal drain valve 2-LV-2708 failed open due to the feedback linkage arm being loose. The feedback linkage arm is attached to the valve stem via a jam nut. The jam nut was apparently loosened by vibration, thus allowing valve stem travel without feedback to the valve positioner.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

The jam nut was apparently loosened by vibration.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF
COMPONENTS WITH MULTIPLE FUNCTIONS

Failure of the valve 2-LV-2708 resulted in the loss of forward flow from the Heater Drain System to the MFPs, resulting in MFP trip, a turbine runback, and subsequent reactor trip.

D. FAILED COMPONENT INFORMATION

Manufacturer: Fisher Controls
Model Number: 12-UR
Serial Number: 6328612

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The following safety system actuations occurred as a result of this event:

Reactor Protection System (RPS) (EIIS:(JC))
Auxiliary Feedwater System (AFW) (EIIS:(BA))

B. DURATION OF SAFETY SYSTEM TRAIN OPERABILITY

Not applicable - there was no safety systems which were rendered inoperable due to this failure.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

This event is bounded by the accident analysis in Chapter 15.2.7 of the Final Safety Analysis Report for Loss of Normal Feedwater. A reactor trip on low Steam Generator (SG) water level in any SG and the initiation of AFW provides the necessary heat removal capability. The reactor automatically tripped on LO-LO SG Level.

Based on the above discussion, it is concluded that this event did not adversely affect the safe operation of CPSES Unit 2 or the health and safety of the public.

IV. CAUSE OF THE EVENT

ROOT CAUSE

The cause of this event was valve 2-LV-2708 failing open. It was determined that valve 2-LV-2708 failed open due to the feedback linkage arm being loose. The feedback linkage arm is attached to the valve stem via a jam nut. The jam nut was found to be loose. The jam nut apparently came loose due to vibration, allowing valve stem travel without feedback to the valve positioner.

V. CORRECTIVE ACTIONS

A. IMMEDIATE CORRECTIVE ACTION

The jam nut was tightened, reattaching the feedback linkage arm to the positioner. Loctite was applied to prevent recurrence. The subject valve was tested to ensure functionality and placed back in service.

A review of the spurious trip of MFP 2-A revealed the probable cause to be a ground on the positive side of the MFP Turbine Trip Solenoid (SV12). During troubleshooting the ground cleared. Terminations at the turbine were inspected for any obvious potential grounds. None were found. Plant computer data was reviewed for any abnormal parameters that may have tripped MFP 2-A. None were found. No other problems were identified and MFP 2-A was subsequently restarted successfully. No further actions were required.

B. CORRECTIVE ACTION TAKEN ON GENERIC CONCERNS IDENTIFIED As A DIRECT RESULT OF THE EVENT

GENERIC CONCERN

The possibility existed that other Fisher Positioner feedback linkage arms, identical to valve 2-LV-2708, could be loose.

CORRECTIVE ACTION

Twenty five (25) Fisher Positioners for Unit 1, Unit 2, and Common, were identified as having feedback linkage arms identical to valve 2-LV-2708. A walkdown was immediately conducted and all twenty five (25) Fisher Positioners were verified to be tight. Additional corrective action was taken and Loctite was placed on jam nuts of these valves.

GENERIC CONCERN

The possibility existed that other Fisher Positioner feedback linkage arms, similar to valve 2-LV-2708, could be loose.

CORRECTIVE ACTION

One hundred forty three (143), Fisher Positioners for Unit 1, Unit 2, and Common, were identified as having feedback linkage arms similar to valve 2-LV-2708. A walkdown was immediately conducted and all one hundred forty three (143) Fisher Positioners were verified to be tight.

VI. PREVIOUS SIMILAR EVENTS

There have been no other previous LERs which dealt with Fisher Positioners feedback linkage arms.

VII. ADDITIONAL INFORMATION

The times listed in the report are approximate and Central Daylight Time.

LER 445/90-025-00 for CPSES Unit 1 reported an event where the mounting nut for a Bailey Positioner on the Feedwater Flow Control valve loosened, allowing the drive rod to separate from the drive arm resulting in a reactor trip. The corrective actions were to initiate a design modification to dampen the vibrations. The corrective actions taken to resolve the previous event would not have prevented this event.

Additionally, as a result of Operating Experience (OE) 5042 Bailey Positioners have been identified for CPSES Unit 1, Unit 2 and common. Appropriate corrective actions are in place and are similar to the actions for the Fisher Positioners.

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Log # TXX-93414
File # 10200
Ref. # 10CFR50.73(a)(2)(i)

TU ELECTRIC

December 15, 1993

William J. Cahill, Jr.
Group Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) - UNIT 2
DOCKET NO. 50-446
MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERING
SAFETY FEATURE
LICENSEE EVENT REPORT 93-011-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 93-011-00 for Comanche Peak Steam Electric Station Unit 2, "Loose Jam Nut on the Positioner Feedback Linkage Arm of Separator Drain Tank Normal Valve Caused an ESF Actuation."

Sincerely,

William J. Cahill, Jr.

OB:tg
Enclosure

cc: Mr. J. L. Milhoan, Region IV
Mr. L. A. Yandell, Region IV

Resident Inspectors, CPSES

400 N. Olive Street L.B. 81 Dallas, Texas 75201

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